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Nuclear Management Company, LLC
Point Beach Nuclear Plant
6610 Nuclear Road
Two Rivers, WI 54241

NPL 2000-0478

November 1, 2000

Document Control Desk
U.S. NUCLEAR REGULATORY COMMISSION
Mail Station P1-137
Washington, DC 20555

10 CFR 50.73

Ladies/Gentlemen:

DOCKET NOS 50-266 AND 50-301
LICENSEE EVENT REPORT 2000-009-00
INITIAL CONDITIONS ASSUMED FOR CONTROL ROOM HABITABILITY DOSE CALCULATION DO NOT MATCH SHIELDING CONFIGURATION
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

Enclosed is Licensee Event Report 2000-009-00 for Point Beach Nuclear Plant, Unit 1 and 2. This report is provided in accordance with 10 CFR 50.73(a)(2)(ii) as, "any event or condition that resulted ...in the nuclear power plant being:...(B) In a condition that was outside the design basis of the plant." This report describes the discovery that, contrary to the assumptions in the post accident control room dose calculation, the portable shielding for the control room observation window and south door is not in place at the beginning of the accident. If the shielding is not moved into place promptly, the possibility exists that the 30 day post accident whole body dose to the control room personnel may exceed the limits specified in 10 CFR 50 Appendix A, GDC 19.

This report contains one new commitment, identified in italics, in the Corrective Actions section.

Please contact us if you require additional information.

Sincerely,

DE Cole

Dan Cole
Manager,
Site Assessment

Enclosure

CWK/tat

cc: NRC Resident Inspector
NRC Regional Administrator
INPO Support Services
NRC Project Manager
PSCW

IE22

NRC FORM 366 (4-95) U.S. NUCLEAR REGULATORY COMMISSION

# LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

APPROVED BY OMB NO. 3150-0104
EXPIRES 04/30/98



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TITLE (4)

Initial Conditions Assumed for Control Room Habitability Dose Calculation Do Not Match Shielding Configuration

| EVENT DATE (5) | | | LER NUMBER (6) | | | REPORT DATE (7) | | | OTHER FACILITIES INVOLVED (8) | |
|---|---|---|---|---|---|---|---|---|---|---|
| MONTH | DAY | YEAR | YEAR | SEQUENTIAL NUMBER | REVISION NUMBER | MONTH | DAY | YEAR | FACILITY NAME | DOCKET NUMBER |
| | | | | | | | | | Unit 2 | 05000301 |
| 10 | 02 | 2000 | 2000 - | 009 - | 00 | 11 | 01 | 2000 | FACILITY NAME | DOCKET NUMBER 05000 |

OPERATING MODE (9): 1

POWER LEVEL (10): 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| | 20.2201(b) | | 20.2203(a)(2)(v) | | 50.73(a)(2)(i) | | 50.73(a)(2)(viii) |
| | 20.2203(a)(1) | | 20.2203(a)(3)(i) | X | 50.73(a)(2)(ii) | | 50.73(a)(2)(x) |
| | 20.2203(a)(2)(i) | | 20.2203(a)(3)(ii) | | 50.73(a)(2)(iii) | | 73.71 |
| | 20.2203(a)(2)(ii) | | 20.2203(a)(4) | | 50.73(a)(2)(iv) | | OTHER |
| | 20.2203(a)(2)(iii) | | 50.36(c)(1) | | 50.73(a)(2)(v) | | Specify in Abstract below or in NRC Form 366A |
| | 20.2203(a)(2)(iv) | | 50.36(c)(2) | | 50.73(a)(2)(vii) | | |

LICENSEE CONTACT FOR THIS LER (12)

| NAME | TELEPHONE NUMBER (Include Area Code) |
|---|---|
| Charles Wm. Krause, Senior Regulatory Compliance Engineer | (920) 755-6809 |

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

| CAUSE | SYSTEM | COMPONENT | MANUFACTURER | REPORTABLE TO EPIX | CAUSE | SYSTEM | COMPONENT | MANUFACTURER | REPORTABLE TO EPIX |
|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | | | |
| | | | | | | | | | |

| SUPPLEMENTAL REPORT EXPECTED (14) | | | | EXPECTED SUBMISSION DATE (15) | MONTH | DAY | YEAR |
|---|---|---|---|---|---|---|---|
| | YES (If yes, complete EXPECTED SUBMISSION DATE). | X | NO | | | | |

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On October 2, 2000, the licensee observed that the post accident control room dose calculation assumed that shielding for the control room observation window and south door is in place at the beginning of the accident to limit the whole body dose to the control room personnel to less than 5 rem during the first 30 days after the accident. The shielding for these locations, although portable, is not installed by procedure until sometime after the initiation of the accident. The dose rate calculations and assumptions have been evaluated against the actual time to reach the procedural step that directs the installation and the actual time it took to put the shielding in place. Based on those determinations, in the event of a design basis accident, it is possible that the 5 rem dose rate could have been exceeded due to increased radiation streaming through the unshielded control room window prior to getting the shielding in place. A one hour NRC notification pursuant to 10 CFR 50.72 for a condition potentially outside the design basis for the plant was made on October 2, 2000. Since February 1999, we have committed to the NRC to maintain an administrative leak rate limit for the containment of one half the Technical Specification limit. Based on that lower permissible leak rate, adequate time exists to install the portable shielding and the 5 rem limit would not be exceeded.

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TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

## Event Description:

While completing an assessment of where to locate a new data base terminal outside the Control Room, an engineer familiar with the Control Room post design basis accident dose calculations observed that the configuration for the portable shielding for the control room door and observation window was not consistent with the assumption in the dose calculations. This condition was documented in a condition report (CR 00-2937). The portable shielding is provided to reduce the dose rate to the operators inside the Control Room from the passing radioactive plume following a large break loss of coolant accident (LOCA). This dose reduction is needed to assure that the whole body dose to the Control Room Operators does not exceed the limits specified in 10 CFR 50, Appendix A, GDC 19. The GDC states that adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of five rem whole body, or its equivalent to any part of the body, for the duration of the accident. The results of the analysis are presented in FSAR Section 14.3.5, "Radiological Consequences of Loss of Coolant Accident."

Both this analysis and the supporting calculation (Calculation 97-0115) assume that the portable shielding is in place at the initiation of the event. This shielding consists of one quarter inch or one half inch lead blankets mounted on mobile frame supports which can be rolled into place at the onset of a design basis accident. Shielding is provided for the south control room door and for the control room observation window. This shielding, although located in close proximity to these features, normally is not in-place so as to not impede access to the observation window or the south door. Directions are included in the plant emergency operating procedures, specifically Step 18 of Emergency Operating Procedure EOP-1, "Loss of Reactor or Secondary Coolant," to move the shielding into place. However, the condition report also documented that the location of several items, including a beverage machine and the platform for the observation window, may impeded the timely deployment of the shielding.

Since the portable shielding would not have been in place at the onset of a design basis loss of coolant accident, as assumed in our analysis, we conservatively declared this configuration deficiency to be a condition that could have resulted in the post accident control room dose being outside the 5 rem total exposure design basis. Accordingly, pursuant to 10 CFR 50.72(b)(1), a one hour ENS notification for a condition outside the design basis was completed at 1800 CDT on October 2, 2000. Pending an engineering review of the calculation and its assumptions, the portable shielding for the control room door and observation window was promptly moved into its post accident positions. This installation, which included relocation of the beverage machine and observation platform and separation of the window shielding bundles from the door bundles, was completed in about one hour.

## Cause:

The purpose of the window in the control room wall is to permit plant personnel and tour groups to observe control room operations without actually entering the control room and disturbing ongoing plant operations. Portable shielding was provided to cover this window in the event of a major design basis accident and thereby reduce the dose rate from a potential radioactive plume outside the control building. We have not identified the basis for why this shielding is not installed immediately after accident

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initiation. Although the dose rate calculation assumes the presence of a radioactive release at the onset of the postulated accident, we believe it was recognized that following a design basis LOCA there would be some amount of time before the post accident radioactive plume would cause actual exposure to the control room personnel, thus allowing time to install the portable shielding. However, the dose analysis and the procedural step for placement of the portable shielding in the EOPs were not compared to ensure that the procedural guidance was consistent with the dose analysis assumptions.

**Corrective Actions:**

The licensee will be conducting an apparent cause evaluation to investigate further the reason for permitting the use of portable shielding versus the shielding configuration assumed in the calculation.

As mentioned in the Event Description, the portable shielding has been moved into position.

*The portable shielding will remain in place until corrective actions are completed that ensure compliance with the five rem whole body dose limit of GDC 19 without the portable shielding in place at the beginning of an accident.*

**Safety Assessment:**

As discussed in the FSAR post accident radiological dose evaluation, the direct dose to the operators in the control room must be below 3.40 rem to provide allowance for 1.37 rem whole body dose from the radionuclides within the control room and 0.225 rem direct radiation from the containment. The current analysis, which assumes the shielding in place, calculates that a control room operator would receive 2.48 rem from the passing plume due to gamma streaming through the shielded control room window. The operators proximity to the window provides the limiting dose case since the streaming dose rate in the vicinity of the door is much less. This analysis is documented in Calculation 97-0115, "Point Beach Nuclear Plant Control Room Internal Dose Rate Due to External Cloud." From the various time step calculations performed as part of this calculation, the dose to an operator positioned in front of the window shielded and unshielded can be determined. The calculation was reevaluated assuming that the shielding was placed in service at 65 minutes into the accident. An operations time validation of the applicable Emergency Operating Procedures, demonstrated that the procedural step which directs the auxiliary operators to install the control room shielding would be reached in approximately 41 minutes. This requires the window shielding to be rolled into position in about 24 minutes. The calculation evaluation determined that under the 65 minute assumption, control room dose increase would be 0.72 rem. Therefore, the whole body dose to the operator for the entire duration of the accident would be 4.78 rem, which is less than the GDC 19 limit of five rem. However, since the actual placement of the shielding on October 2 exceeded 45 minutes, we have conservatively concluded that the design basis five rem limit could, given the postulated accident, potentially have been exceeded. Therefore, we are submitting this report.

It should be noted that the dose calculation and analysis assumes that the containment leakage rate is at the Technical Specification limit of 0.4% by weight per 24 hours for

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the first 24 hours and 0.2% per day thereafter as required by Regulatory Guide 1.4. In a letter dated February 24, 1999, PBNP has committed to the NRC to administratively limit the containment leakage rate to half the Technical Specification values. This commitment is also documented in Sections 6.2 and 9.8 of the FSAR. Using this administrative leak rate limit results in an overall factor of two reduction for the control room dose due to the containment leakage. Accordingly, although the potential existed to exceed a design basis post accident total operator dose limit, the actual consequences of this event given our low containment leak rate experience, would be significantly less than determined in our conservative calculation. Therefore, there was no impact as a result of this event on the health and safety of the public and the safety significance for the plant staff was minimal.

This event did not involve the failure of a component or system to perform its safety related function; therefore, this event does not constitute a safety system functional failure.

## System and Component Identifiers:

The Energy Industry Identification System component function identifier for each component/system referred to in this report are as follows:

| Component/System | Identifier |
|---|---|
| Control Building Control Complex | NA |
| Reactor Containment Building | NH |
| Control Building Environmental Control System | VI |
| Door | DR |

## Similar Occurrences:

A review of recent LERs (past two years) identified no other events which involved radiological dose assessment changes as a result of configuration control concerns.